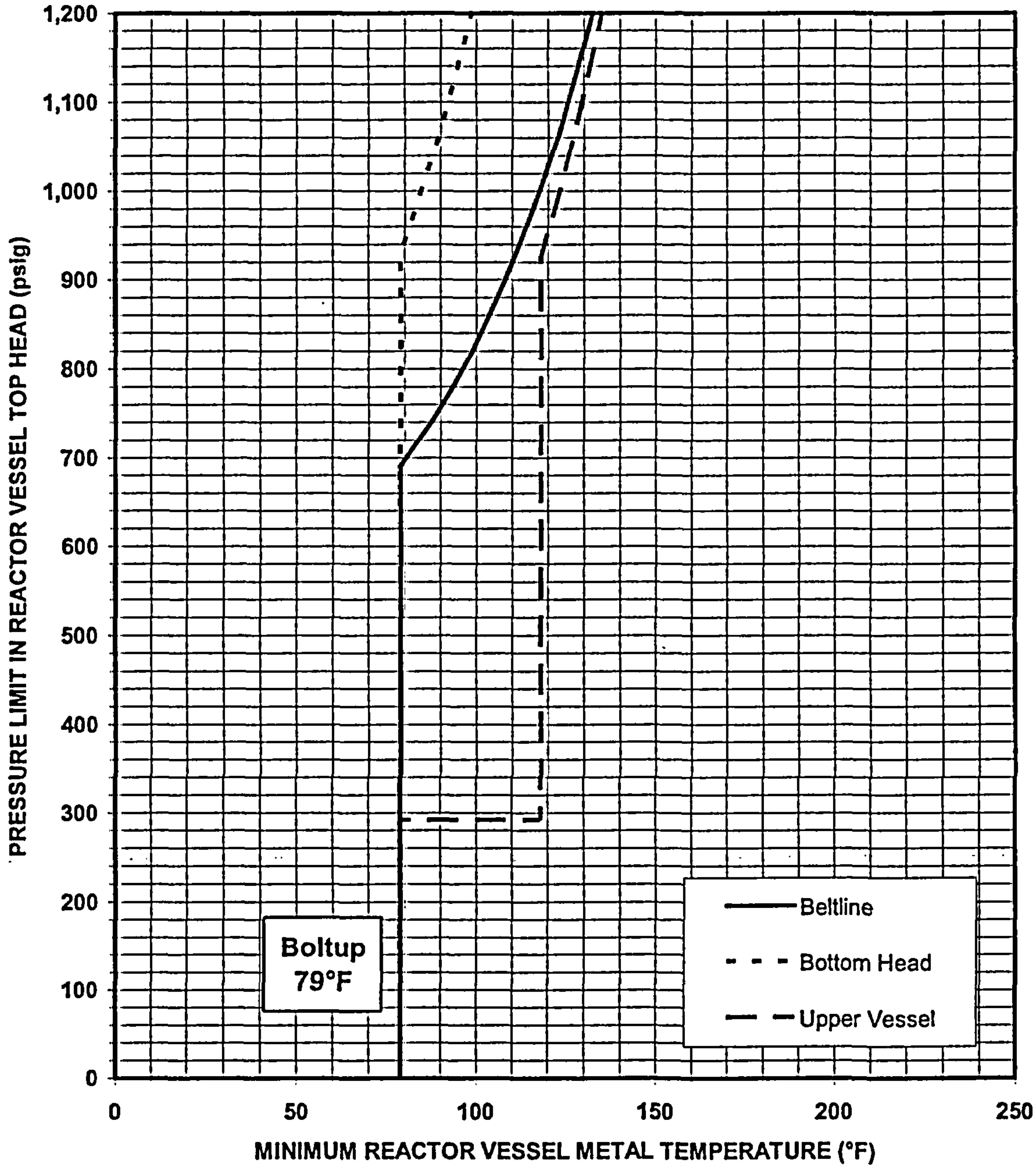
BASES

SECTION		PAGE
INSTRUMENTATION (Continued)		
	Remote Shutdown Monitoring Instrumentation and Controls	B 3/4 3-5
	Accident Monitoring Instrumentation	B 3/4 3-5
	Source Range Monitors	B 3/4 3-5
	Traversing In-Core Probe System	B 3/4 3-5
3/4.3.8	DELETED	B 3/4 3-7
3/4.3.9	FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION	B 3/4 3-7
	Figure B3/4 3-1 Reactor Vessel Water Level	B 3/4 3-8
3/4.3.10	MECHANICAL VACUUM PUMP TRIP INSTRUMENTATION	B 3/4 3-9
3/4.4 REACTOR COOLANT SYSTEM		
3/4.4.1	RECIRCULATION SYSTEM	B 3/4 4-1
3/4.4.2	SAFETY/RELIEF VALVES	B 3/4 4-2
3/4.4.3	REACTOR COOLANT SYSTEM LEAKAGE	
	Leakage Detection Systems	B 3/4 4-3
	Operational Leakage	B 3/4 4-3
3/4.4.4	CHEMISTRY	B 3/4 4-3
3/4.4.5	SPECIFIC ACTIVITY	B 3/4 4-4
3/4.4.6	PRESSURE/TEMPERATURE LIMITS	B 3/4 4-5
	Table B3/4.4.6-1 Reactor Vessel Toughness	B 3/4 4-7
	Figure B3/4.4.6-1 Fast Neutron Fluence (E>1Mev) at (1/4)T as a Function of Service life	B 3/4 4-8
	Table B3/4.4.6-2 Numeric Values for Pressure/Temperature Limits	B 3/4 4-9

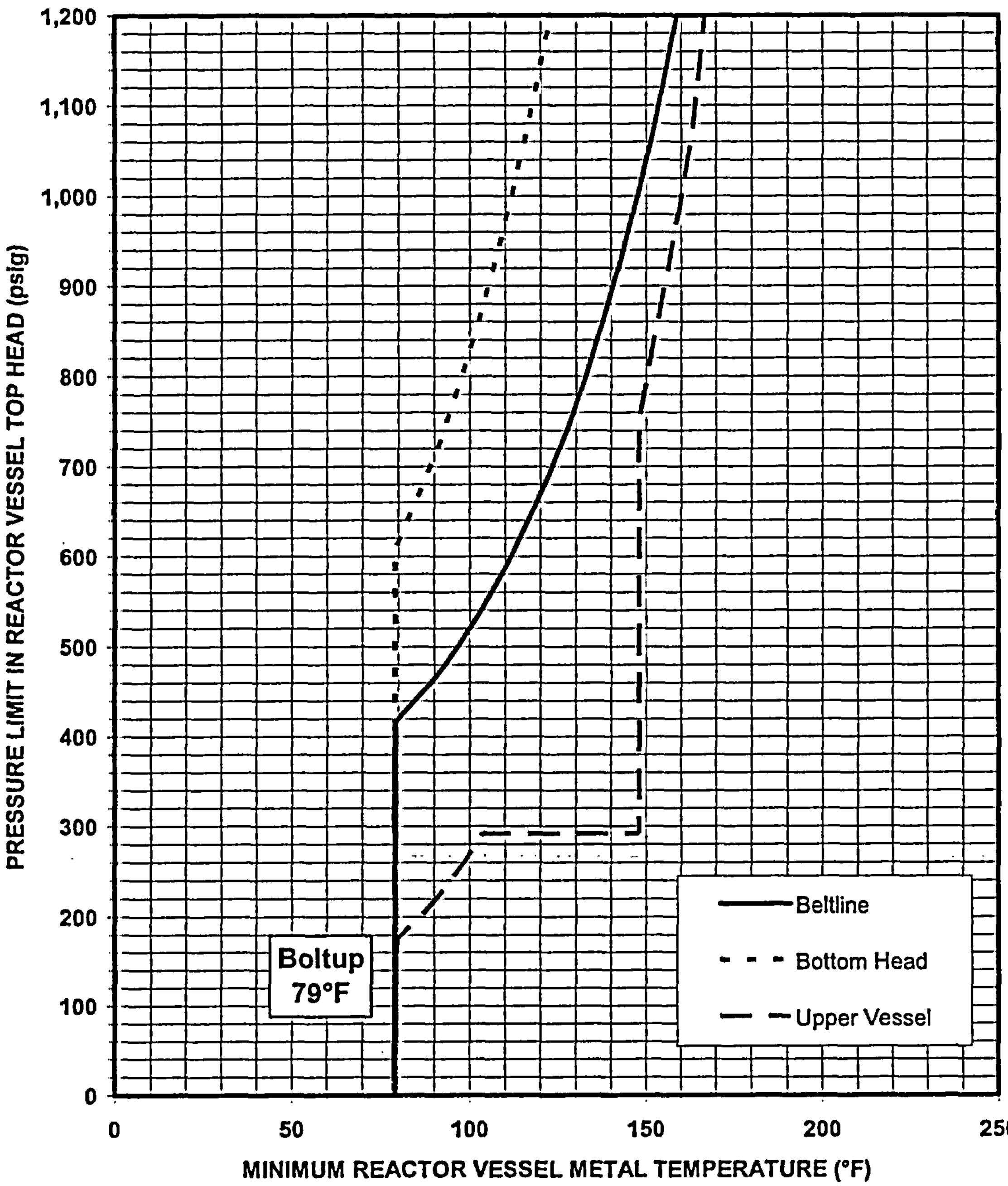
Figure 3.4.6.1-1
Hydrostatic Pressure and Leak Tests Pressure/Temperature Limits – Curve A



All system leakage and hydrostatic pressure tests performed during the service life of the pressure boundary in compliance with ASME Code Section XI.

This figure is valid for 32 EFPY of operation.

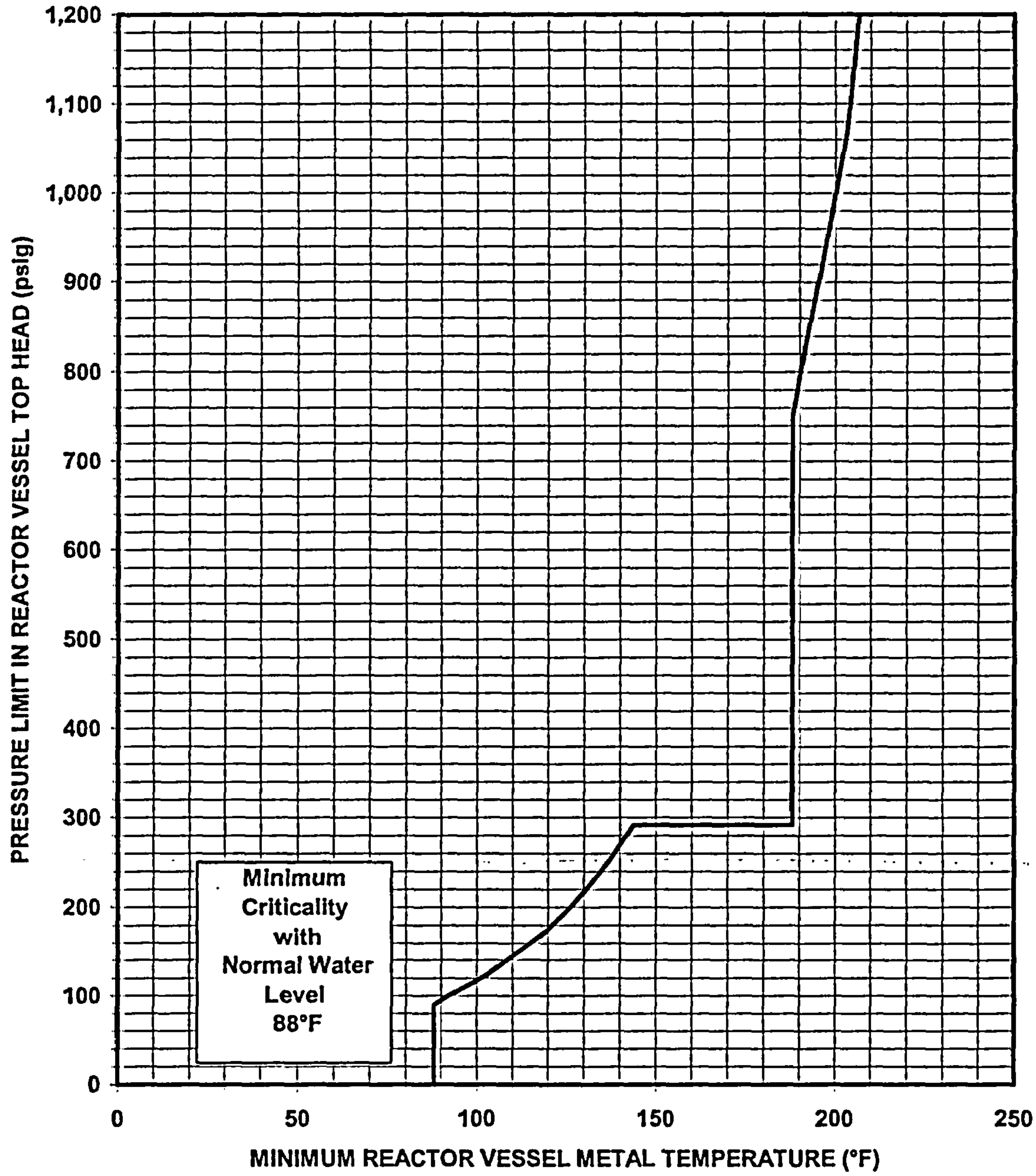
Figure 3.4.6.1-2
Non-Nuclear Heatup and Cooldown Pressure/Temperature Limits – Curve B



All heatup and cooldowns that are performed when the reactor is not critical at the normal heatup and cooldown rate.

This figure is valid for 32 EFPY of operation.

Figure 3.4.6.1-3
Core Critical Heatup and Cooldown Pressure/Temperature Limits – Curve C



All heatup and cooldowns that are performed when the reactor is critical at the normal heatup and cooldown rate.

This figure is valid for 32 EFPY of operation.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section (3.9) of the UFSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation. Specifically the average rate of change of reactor coolant temperature during normal heatup and cooldown shall not exceed 100°F during any 1-hour period.

The operating limit curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 are derived from the fracture toughness requirements of 10 CFR 50 Appendix G and ASME Code Section XI, Appendix G and ASME Code Cases N-588 and N-640. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis for compliance are more fully discussed in UFSAR Chapter 5, Paragraph 5.3.1.5, "Fracture Toughness." Tabulated values for the P-T curves are shown in Table B 3/4.4.6-2.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of some of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, nickel content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Material". The pressure/temperature limit curves, Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3, includes an assumed shift in RT_{NDT} for the end of life fluence.

The fluence in Bases Figure B 3/4.4.6-1 was determined using methodology described in NRC-approved General Electric Nuclear Energy Licensing Topical Report NEDC-32983P-A. This methodology is consistent with the guidance in Regulatory Guide 1.190, Rev. 0, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, irradiated flux wires installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the flux wires and vessel inside radius are essentially identical, the irradiated flux wires can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 shall be adjusted, as required, on the basis of the flux wire data and recommendations of Regulatory Guide 1.99, Rev. 2.

BASES TABLE B 3/4.4.6-1

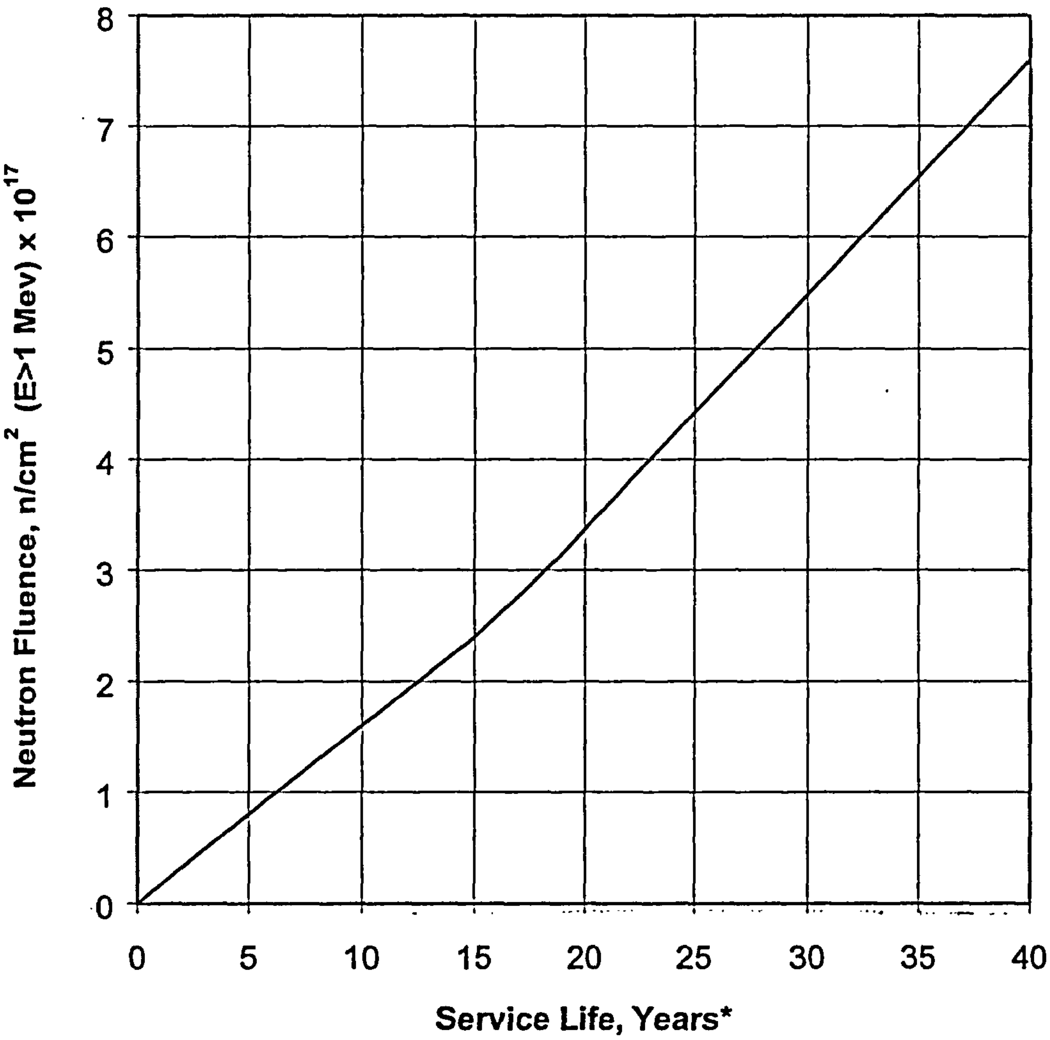
REACTOR VESSEL TOUGHNESS

BELTLINE COMPONENT	WELD SEAM I.D. OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU(%)	Ni(%)	HIGHEST RT_{NDT}(°F)	ΔRT_{NDT} + MARGIN (°F)	PREDICTED EOL UPPER SHELF (FT-LBS)	MAX. EOL RT_{NDT}(°F)
Plate	SA-533 GR B CL.1	5K3025-1	.15	0.71	+19	56	66	75
Weld	Vert. seams for shells 4&5	D53040/ 1125-02205	0.081	0.611	-30	78	118	48

NOTE: * These values are given only for the benefit of calculating the end-of-life (EOL) RT_{NDT}.

NON-BELTLINE COMPONENT	MT'L TYPE OR WELD SEAM I.D.	HEAT/SLAB OR HEAT/LOT	HIGHEST REFERENCE TEMPERATURE RT_{NDT} (°F)
Shell Ring Connected to Vessel Flange	SA 533, GR.B, Cl.1	All Heats	+19
Bottom Head Dome	SA 533, GR.B, Cl.1	All Heats	+30
Bottom Head Torus	SA 533, GR.B, Cl.1	All Heats	+30
LPCI Nozzles(1)	SA 508, Cl.2,	All Heats	-20
Top Head Torus	SA 533, GR.B, Cl.1	All Heats	+19
Top Head Flange	SA 508, Cl.2	All Heats	+10
Vessel Flange	SA 508, Cl.2	All Heats	+10
Feedwater Nozzle	SA 508, Cl.2	All Heats	-20
Weld Metal	All RPV Welds	All Heats	0
Closure Studs	SA 540, GR.B, 24	All Heats	Meet 45 ft-lbs & 25 mils lateral expansion at +10°F

(1) The design of the Hope Creek vessel results in these nozzles experiencing a predicted EOL fluence at 1/4T of the vessel thickness of 3.3×10^{17} n/cm^2. Therefore, these nozzles are predicted to have an EOL RT_{NDT} of +29°F.



LOWER - INTERMEDIATE SHELL FAST NEUTRON FLUENCE (E>1 MeV)
AT 1/4 T AS A FUNCTION OF SERVICE LIFE*

Bases Figure B 3/4.4.6-1

*At 80% capacity factor (40 years = 32 EFPY)

BASES TABLE B 3/4.4.6-2

Numeric Values for Pressure/Temperature Limits

Figure 3.4.6.1-1, Curve A

Bottom Head	
Temperature (°F)	Pressure (psig)
79	0
79	929
88	1040
90	1068
92	1097
94	1126
96	1157
98	1190
100	1223

Upper Vessel	
Temperature (°F)	Pressure (psig)
79.0	0
79.0	292
118.0	292
118.0	925
123.0	996
128.0	1074
133.0	1161
138.0	1257

Beltline	
Temperature (°F)	Pressure (psig)
79.0	0
79.0	691
88.0	743
93.0	777
98.0	814
103.0	855
108.0	900
113.0	950
118.0	1,005
123.0	1,065
128.0	1,133
133.0	1,207

Figure 3.4.6.1-2, Curve B

Bottom Head	
Temperature (°F)	Pressure (psig)
79	0
79	606
88	690
92	732
96	778
100	827
104	881
108	939
112	1002
116	1070
120	1144
124	1224

Upper Vessel	
Temperature (°F)	Pressure (psig)
79.0	0
79.0	50
79.0	75
79.0	90
79.0	100
79.0	125
79.8	175
86.6	202
90.6	220
96.6	250
98.4	260
102.6	285
103.7	292
148.0	292
148.0	740
148.0	745
148.0	750
151.6	830
155.8	910
159.7	990
163.3	1070
165.5	1150
167.5	1230

Beltline	
Temperature (°F)	Pressure (psig)
79.0	0
79.0	416
88.0	455
93.0	480
98.0	508
103.0	538
108.0	572
113.0	610
118.0	651
123.0	697
128.0	747
133.0	803
138.0	864
143.0	932
148.0	1,008
153.0	1,091
158.0	1,183
163.0	1,284

BASES TABLE B 3/4.4.6-2 (continued)

Numeric Values for Pressure/Temperature Limits

Figure 3.4.6.1-3, Curve C

Temperature (°F)	Pressure (psig)
88.0	0
88.0	50
88.0	75
88.0	90
92.0	100
103.4	125
119.8	175
126.6	202
130.6	220
136.6	250
138.4	260
142.6	285
143.7	292
188.0	292
188.0	740
188.0	745
188.0	750
191.6	830
195.8	910
199.7	990
203.3	1070
205.5	1150
207.5	1230